U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

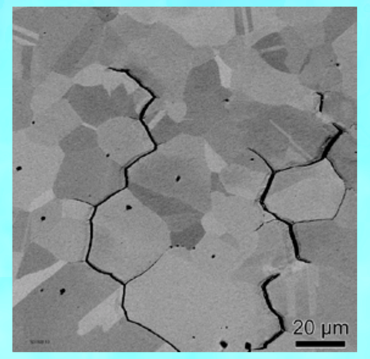
Office of Nuclear Regulatory Research

U.S. Nuclear Regulatory Commission-Sponsored Research on Environmentally Assisted Cracking

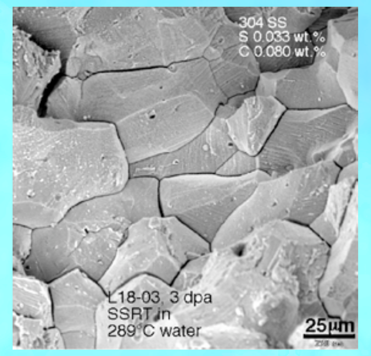
Introduction

Environmentally assisted cracking (EAC) is a degradation mechanism that can lead to the brittle failure of ductile engineering alloys. Two forms of EAC that are relevant to commercial nuclear power reactors and reactor coolant systems are irradiation-assisted stress corrosion cracking (IASCC) and primary water stress-corrosion cracking (PWSCC). A third form of EAC that may be significant under a limited range of conditions is low-temperature crack propagation (LTCP):

- Irradiation-assisted stress corrosion cracking
  - Observed in light-water reactor stainless steel components
  - Susceptibility increases with increased neutron fluence
- Primary water stress corrosion cracking
  - Observed in Nickel alloys in pressurized-water reactor (PWR) systems
  - Fabrication processes and welding increase susceptibility
- Low-temperature crack propagation
  - Not identified in operating reactors
  - Observed in nickel-base alloys at low temperature in simulated PWR environments



Metallurgical section showing intergranular primary water stress corrosion cracking



Fracture surface showing intergranular features as a result of irradiation-assisted stress corrosion cracking

Objectives of the NRC-Sponsored Research

- Understand environmental, metallurgical, and fabrication-related variables that affect EAC susceptibility and crack growth rates.
- Develop information to support regulatory guidance and decisions in applicable areas.

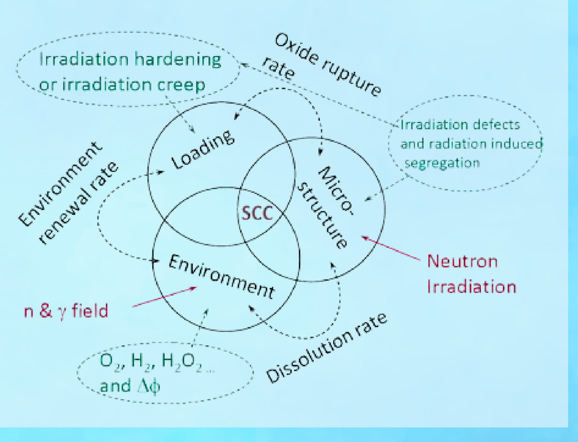
Current Research Efforts

- Argonne National Laboratory (ANL)
  - Irradiation-assisted stress corrosion cracking
  - Primary water stress corrosion cracking
- Pacific Northwest National Laboratory (PNNL)
  - Primary water stress corrosion cracking
  - Low-temperature crack propagation

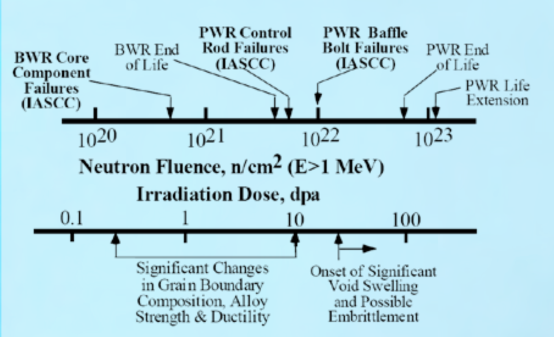
Irradiation-Assisted Stress Corrosion Cracking

Significance for Nuclear Power Plants

- As nuclear plants age and neutron fluence increases, nuclear-grade stainless steels can become susceptible to irradiation-assisted stress corrosion cracking.
- Some safety-significant structural components such as the boiling-water reactor (BWR) top guide, shroud, and core plate) may exhibit IASCC at high fluence near the end of life.



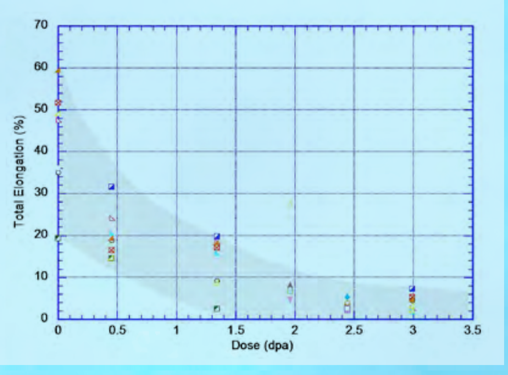
Factors that influence irradiation-assisted stress corrosion cracking



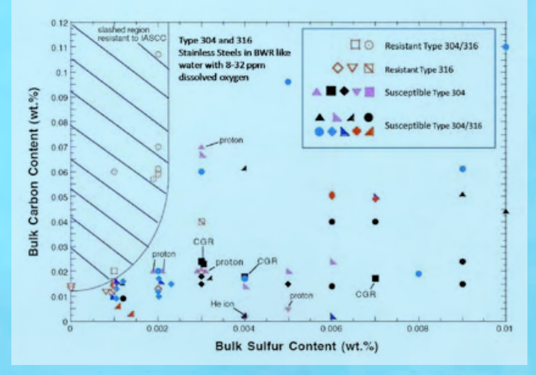
Relationship between neutron fluence, dose, metallurgical changes, and irradiation-assisted stress corrosion cracking

IASCC Factors

- Loading: stress intensity, stress intensity factor gradient, loading rate
- Environment: temperature, electrochemical potential, dissolved oxygen concentration, flow rate, conductivity, pH
- Microstructure: composition, segregation characteristics, stacking fault energy, precipitation, metallurgical condition, yield strength
- Irradiation: dose, temperature, flux



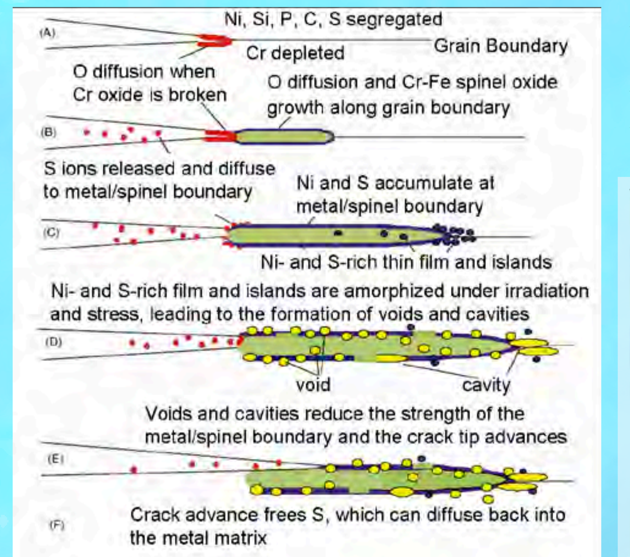
Effect of dose on ductility of 300 series stainless steels



Effect of carbon and sulfur content on the IASCC susceptibility of types 304 and 316 stainless steels

Possible IASCC Mechanisms

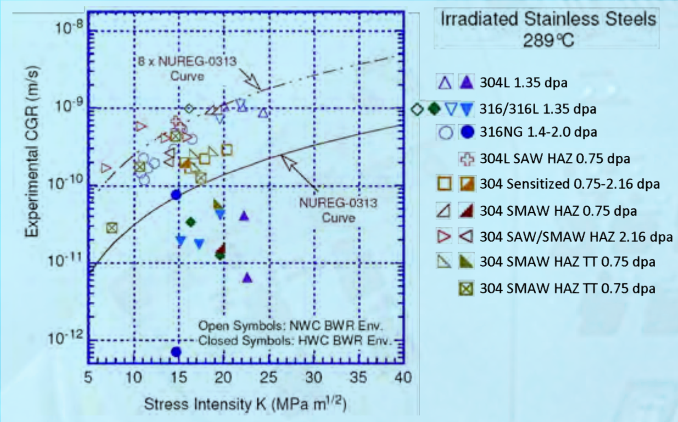
- Radiation-induced segregation (RIS), the spatial redistribution of elements at defect sinks under irradiation, which leads to element depletion or enrichment at grain boundaries
- Hardening and loss of ductility due to irradiation defects
- Localized plastic flow due to cleared dislocation channels
- Void swelling, irradiation creep
- Corrosion potential change due to radiolysis



Proposed model for irradiation-assisted stress corrosion cracking

Current NRC-Sponsored IASCC Research

- Evaluate the effectiveness of strees corrosion cracking mitigation methods
  - hydrogen water chemistry
  - grain boundary engineering
- Evaluate crack growth rate (CGR) models for light-water reactors
- Evaluate the causes, mechanisms, and effects of EAC on light-water reactor internals
- Investigate radiation embrittlement under PWR conditions
- Study synergistic effects associated with radiation and thermal embrittlement in cast austenitic stainless steels



Effect of normal water chemistry (NWC) and hydrogen water chemistry (HWC) on measured IASCC growth rates for types 304 and 316 stainless steels

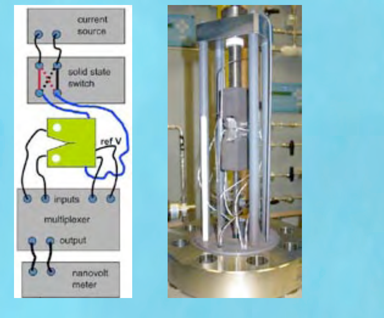
Primary Water Stress Corrosion Cracking

Significance for Nuclear Power Plants

- Nickel-base alloys and weld metals used for primary pressure boundary components may be susceptible to primary water stress corrosion cracking (PWSCC)
- PWSCC is a significant safety concern for pressurized water reactors
  - Potential for reactor pressure boundary leaks and boric acid corrosion of low alloy steels
  - Possible loss of coolant accident

Removed section of the Davis-Besse reactor head (left) showing extensive boric acid corrosion of the low alloy steel. PWSCC of the control rod drive mechanism nozzle Alloy 182 J-groove weld (right) allowed reactor coolant to contact and corrode the low alloy steel reactor vessel head.


Laboratory test system for measuring PWSCC growth rates



Schematic of crack growth instrumentation (left) and actual specimen in test fixture (right)

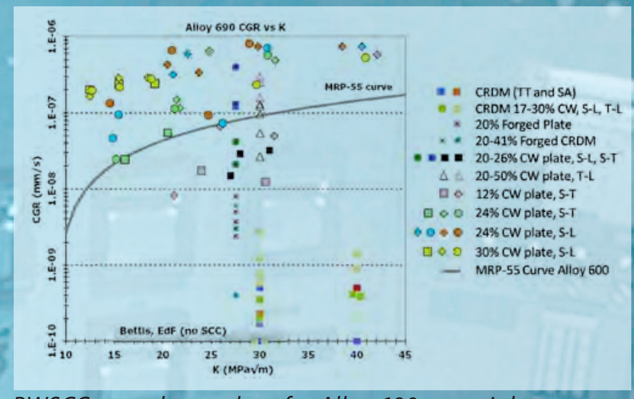
PWSCC Factors

- Alloy Composition: Chromium concentration and chemical homogeneity
- Metallurgical Factors: grain size, banding, carbide distribution
- Fabrication Processes: residual stresses and strains, heat-affected zones, weld metal composition, microstructure, orientation, and dissimilar metal dilution zones
- Environment: temperature, hydrogen concentration, electrochemical potential

Current NRC-Sponsored PWSCC Research

- Measure crack growth rates in welded materials
  - Effect of composition
  - Effect of dilution zones
- Determine crack propagation in heat-affected zones
- Evaluate the effects of processing and crack orientation in Alloy 690
  - Materials processing by forging and rolling
  - Control rod drive mechanism tubing
- Evaluate the efficacy of mitigation methods


PWSCC crack growth rate specimens machined Alloy 690 CRDM tubing material



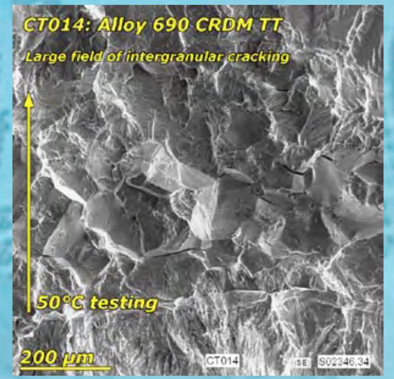
PWSCC growth rate data for Alloy 690 materials. Cold work, microstructural banding, and high stress intensities can increase crack growth rates.

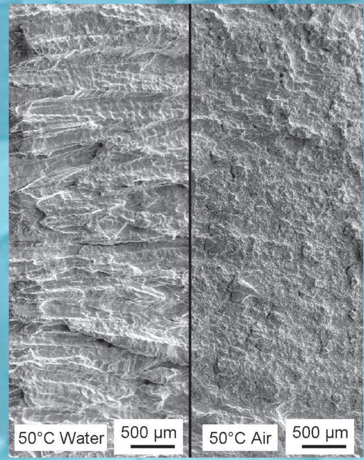
Low-Temperature Crack Propagation

Significance for Nuclear Power Plants

- Intergranular cracking of nickel-alloys has been observed in simulated PWR water chemistry at 50°C
- Crack propagation during reactor shutdown or when returning to high temperatures is not understood

Intergranular fracture surface of Alloy 690 after testing at low temperature

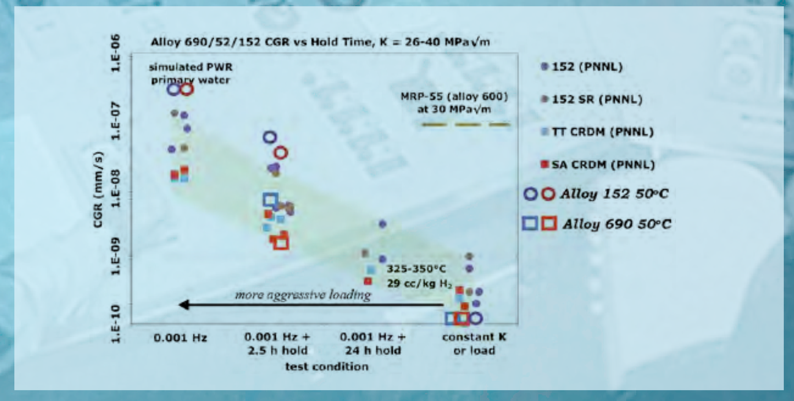




Intergranular fracture surface of Alloy 690 after testing at low temperature

Current NRC-Sponsored LTCP Research

- Evaluate crack growth rate testing of Alloy 690 and its weld metals at 50°C
- Examine metallurgical and environmental factors that promote LTCP



Effect of loading condition on crack growth rates. Lower rates are observed at low temperatures under constant loading conditions.

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